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April 9, 2014

AEP-NRC-2014-24
10 CFR 50.90

Docket Nos.: 50-315
50-316

U. S. Nuclear Regulatory Commission
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Donald C. Cook Nuclear Plant Units 1 and 2
License Amendment Request Regarding a Change to the Reactor Coolant System Pressure and Temperature Limits

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, is submitting a request for an amendment to the Technical Specifications (TS) for CNP Unit 1 and Unit 2.

This application for amendment to the CNP Unit 1 and Unit 2 TS proposes to revise TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits", to address an issue regarding the applicability of Figures 3.4.3-1 "Reactor Coolant System Pressure versus Temperature Limits – Heatup Limit, Criticality Limit, and Leak Test Limit (Applicable for service period up to 32 EFPY)" and 3.4.3-2 "Reactor Coolant System Pressure versus Temperature Limits – Various Cooldown Rates Limits (Applicable for service period up to 32 EFPY)" during vacuum fill operations of the Reactor Coolant System (RCS). Specifically, the curves are being modified to extend to -14.7 pounds per square inch gage to bound the RCS conditions required to support vacuum fill operations.

TS Figures 3.4.3-1 and 3.4.3-2 provide the RCS pressure versus temperature limits for various modes of reactor operation. These curves specify safe zones of reactor operation under varying RCS P/T conditions.

Enclosure 1 to this letter provides an affirmation statement. Enclosure 2 is an evaluation of the proposed change to Section 3.4.3 of the Unit 1 and Unit 2 TSs. Enclosures 3 and 4 are marked up copies of the applicable Unit 1 and Unit 2 TS pages, respectively. Enclosures 5 and 6 are marked up copies of the applicable Unit 1 and Unit 2 TS Bases pages, respectively, provided for information purposes.

I&M requests review and approval of this application by September 5, 2014, in order to incorporate these changes into the CNP refueling outage schedule. The license amendment will be implemented within 60 days of U.S. Nuclear Regulatory Commission approval. This would allow for the use of vacuum fill of the RCS during the currently scheduled Unit 1 Cycle 26 refueling outage in the fall of 2014 and the Unit 2 Cycle 22 refueling outage in the spring of 2015, as well as subsequent refueling outages.

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Copies of this letter and its enclosures are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

There are no new regulatory commitments made in this letter. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Quinton S. Lies

Q. Shane Lies
Engineering Vice President
Indiana Michigan Power Company

JMT/amp

Enclosures:

1. Affirmation
2. Safety Evaluation of Proposed Changes to Unit 1 and Unit 2 Technical Specification 3.4.3
3. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages Marked to Show Proposed Changes
4. Donald C. Cook Nuclear Plant Unit 2 Technical Specification Pages Marked to Show Proposed Changes
5. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Bases Pages Marked to Show Proposed Changes for Information
6. Donald C. Cook Nuclear Plant Unit 2 Technical Specification Bases Pages Marked to Show Proposed Changes for Information

c: J. T. King- MPSC
MDEQ - RMD/RPS
NRC Resident Inspector
C. D. Pederson, NRC Region III
T. J. Wengert - NRC Washington DC
A. J. Williamson, AEP Ft. Wayne, w/o attachments

Enclosure 1 to AEP-NRC-2014-24

AFFIRMATION

I, Q. Shane Lies, being duly sworn, state that I am Engineering Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the U. S. Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

Quinton S. Lies

Q. Shane Lies
Engineering VicePresident

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 9th DAY OF April 2014

Notary Public

My Commission Expires 01/21/2018

Enclosure 2 to AEP-NRC-2014-24

Safety Evaluation of Proposed Changes to Unit 1 and Unit 2 Technical Specification 3.4.3

This license amendment request (LAR) is to amend Operating License Numbers DPR-58 and DPR-74 for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, respectively. Documents referenced in this enclosure are identified in Section 6.0 of this enclosure.

1.0 SUMMARY DESCRIPTION

Indiana Michigan Power Company (I&M), licensee for CNP Unit 1 and Unit 2, requests an amendment to the CNP Unit 1 Operating License DPR-58 and Unit 2 Operating License DPR-74 by incorporating the proposed change for the CNP Units 1 and 2 Technical Specifications (TS). The proposed change is a request to revise TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits" for CNP Units 1 and 2.

I&M requests review and approval of this application by September 5, 2014, in order to incorporate these changes into CNP's refueling outage schedule. The license amendment will be implemented within 60 days of the issuance of the license amendment. This would allow for the use of vacuum fill of the Reactor Coolant System (RCS) during the currently scheduled Unit 1 Cycle 26 refueling outage in the fall of 2014 and the Unit 2 Cycle 22 refueling outage in the spring of 2015, as well as subsequent refueling outages.

2.0 DETAILED DESCRIPTION

2.1 Proposed Change

The proposed amendment would revise the CNP Unit 1 and Unit 2 TS 3.4.3, changing Figures 3.4.3-1 “Reactor Coolant System Pressure versus Temperature Limits – Heatup Limit, Criticality Limit, and Leak Test Limit (Applicable for service period up to 32 EFPY)” and 3.4.3-2 “Reactor Coolant System Pressure versus Temperature Limits – Various Cooldown Rates Limits (Applicable for service period up to 32 EFPY)”. The proposed changes to these figures would include extending the y-axis (ordinate) below 0 pounds per square inch gauge (psig), as well as extending the curves to -14.7 psig. These changes provide the graphical representation of vacuum conditions, which would be encountered during vacuum fill evolutions

In addition, following approval of the proposed amendment, I&M would revise the CNP Unit 1 and Unit 2 TS Bases Page B 3.4.3-1 by inserting a clarifying statement that RCS vacuum fill is an acceptable condition since the resulting P/T combination is located in the region to the right and below the operating limits provided in Figures 3.4.3-1 and 3.4.3-2.

Enclosure 3 contains the existing Unit 1 TS Pages 3.4.3-3 and 3.4.3-4 marked up to show the proposed changes to Figure 3.4.3-1 and Figure 3.4.3-2, respectively. Enclosure 4 contains the existing Unit 2 TS Pages 3.4.3-3 and 3.4.3-4 marked up to show the proposed changes to Figure 3.4.3-1 and Figure 3.4.3-2, respectively. New text on these pages is enclosed in single line border. Changes to the figures are marked with revision lines in the right hand margin. Enclosures 5 and 6 contain the existing TS Bases Page B 3.4.3-1, for Unit 1 and Unit 2, respectively, marked up to show the proposed change for information purposes only. New clean Unit 1 and Unit 2 TS pages, with proposed changes incorporated, will be provided to the U. S. Nuclear Regulatory Commission (NRC) Licensing Project Manager when requested.

2.2 Background

Reactor Coolant System Description

The CNP Unit 1 and Unit 2 RCS consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam generator (SG). The system also includes a pressurizer, connecting piping, pressurizer safety and relief valves, and relief tank, necessary for operational control.

During operation, the reactor coolant pumps (RCP) circulate pressurized water through the reactor vessel and the four reactor coolant loops. The RCS provides a boundary for containing the coolant under operating temperature and pressure conditions. During transient operation, the system's heat capacity attenuates thermal transients generated by the core or SGs.

By appropriate selection of the inertia of the RCPs, the thermal-hydraulic effects are reduced to a safe level during the pump coast down, which would result from a loss-of-flow situation. The layout of the system assures natural circulation capability following a loss-of-flow to permit decay heat removal without overheating the core. Part of the system's piping serves as part of the emergency core cooling system to deliver cooling water to the core during a loss of coolant accident.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained through the use of electrical heaters and sprays. Steam can either be formed by the heaters, or condensed by a pressurizer spray, to minimize pressure variations due to contraction and expansion of the coolant. Spring-loaded safety valves and power-operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank (PRT), where the discharged steam is condensed and cooled by mixing with water.

Reason for Requested Change

TS Figures 3.4.3-1 and 3.4.3-2 provide the RCS pressure versus temperature limits for various modes of reactor operation. These curves specify safe zones of reactor operation under varying RCS P/T conditions.

Since the pressure axis for these curves terminates at 0 psig, operation of the RCS in a pressure condition less than 0 psig (e.g. RCS vacuum fill conditions) is not enveloped per the current TS defined P/T conditions.

The proposed changes to TS Figures 3.4.3-1 and 3.4.3-2 are necessary to reflect RCS pressure conditions experienced during RCS vacuum fill operation.

3.0 TECHNICAL EVALUATION

3.1 Technical Basis for Change

TS Figure 3.4.3-1, Reactor Coolant System Pressure versus Temperature Limits – Heatup Limit, Criticality Limit, and Leak Test Limit (Applicable for service period up to 32 EFPY) and Figure 3.4.3-2, Reactor Coolant System Pressure versus Temperature Limits – Various

Cooldown Rates Limits (Applicable for service period up to 32 EFPY) both limit the "Acceptable Operation" realm of the curve to a RCS pressure of 0 psig or greater.

Following the installation of Reactor Coolant Vacuum Refill System (RCVRS) connections to the Unit 1 and Unit 2 RCS, the RVCRS connections are used during each refueling outage, and the RCS is subjected to a pressure less than 0 psig (i.e. vacuum) via a temporary modification that is controlled by plant procedures.

WCAP-15878, "D.C. Cook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation for 40 Years and 60 Years, Revision 0 (December 2002)," provides the basis for the P/T limit curves for Unit 1 and WCAP-15047, "D. C. Cook Unit 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation, Revision 2 (May 2002)," provides the basis for the P/T limit curves for Unit 2. Both WCAP-15878 and WCAP-15047 were verified to have appropriately incorporated the NRC-approved methodologies contained within WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves (January 1996)." The Westinghouse methodology is an approved generic methodology for generating P/T limits. The Westinghouse methodology provides a discussion of several P/T limit curve generation approaches and equations to generate the necessary P/T limit curves. This NRC-approved methodology does not preclude P/T limits from being revised to include a pressure less than 0 psig and therefore supports the extension of the P/T curves to -14.7 psig.

Westinghouse SECL-96-226, Reactor Coolant Vacuum Refill System Final SECL, Revision 0 (in its entirety) assessed the structural integrity of the Unit 1 and Unit 2 reactor vessels, SGs, RCPs, RCP seals, piping and components to determine if any detrimental impact would result from vacuum refill of the RCS. Westinghouse SECL-96-226 concluded that the Unit 1 and Unit 2 RCS components and piping are sufficiently robust and will not be adversely impacted by the negative pressure present during the vacuum refill process. Westinghouse SECL-96-226 stated that operation of the RCVRS will have no effect on RCS components and piping, and that the integrity of the RCS pressure boundary will remain unaltered.

The proposed heatup and cooldown limitation curves were generated in accordance with the fracture toughness requirements of 10 CFR 50 Appendix G, and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) code, Section XI, Appendix G 1995 edition with 1996 Addenda. The proposed heatup and cooldown limitation curves were established in compliance with the methodology used to calculate and predict effects of radiation on embrittlement of reactor pressure vessel beltline materials. Use of this methodology provides compliance with the intent of 10 CFR 50 Appendix G and provides margins of safety that ensure non-ductile failure of the reactor pressure vessel will not occur. The proposed heatup and cooldown limitation curves prohibit operation in regions where it is possible for non-ductile failure of carbon and low alloy RCS materials to occur: hence, the primary coolant pressure boundary integrity will be maintained throughout the limit of applicability of the curves.

Westinghouse has provided to I&M, correspondence MCOE-LTR-14-17, "Applicability of the Pressure-Temperature (P/T) Limit Curve Figures During Vacuum Refill of the RCS in Mode 5 for Westinghouse and CE NSSS Plants, Revision 0," which concludes that vacuum refill of the Unit 1 and Unit 2 RCS in Mode 5 does not violate the 10 CFR 50, Appendix G P/T requirements for the reactor vessel and remains in compliance with ASME Appendix G limits. Additionally, the NRC-approved methodologies contained within WCAP-14040-NP-A do not preclude the P/T limits from being revised to include a pressure less than 0 psig.

3.2 Conclusions

The proposed changes are in accordance with the NRC approved methodology for developing P/T curves in WCAP-14040-NP-A, which is the basis for the CNP plant specific P/T curves. Additionally, proposed changes to the CNP P/T limits are in accordance with Appendix G to Section XI of the ASME Code and satisfy the requirements of Appendix G to 10 CFR Part 50. Lastly, the proposed changes will have no adverse impact to the structural integrity of the RCS.

4.0 REGULATORY EVALUATION

TS Figures 3.4.3-1 and 3.4.3-2 provide the RCS pressure versus temperature limits for various modes of reactor operation. These curves specify safe zones of reactor operation under varying RCS P/T conditions.

Since the pressure axis for these curves terminates at 0 psig, operation of the RCS in a pressure condition less than 0 psig (e.g. RCS vacuum fill conditions) is not enveloped per the current TS defined P/T conditions

The proposed changes were developed in accordance with 10 CFR 50 Appendix G and ASME B&PV Code Section XI Appendix G, 1995 Edition with 1996 Addenda.

4.1 Applicable Regulatory Requirements/Criteria

<u>Technical Specifications</u>

The proposed changes to TS Figures 3.4.3-1 and 3.4.3-2 are necessary to reflect RCS pressure conditions experienced during RCS vacuum fill operation. With this change, the TS will continue to assure that the necessary quality of this system and its components is maintained and the limiting conditions for operation of this system will continue to be met. Therefore, the requirements of 10 CFR 50.36 continue to be met with the changes proposed in this license amendment request.

<u>10 CFR Appendix G and ASME B&PV Code Section XI Appendix G, 1995 Edition with 1996 Addenda.</u>

10 CFR 50 Appendix G, by reference to ASME B&PV Code Section XI Appendix G, 1998 Edition with 2000 Addenda, specifies fracture toughness and testing requirements for reactor vessel materials. Use of the NRC approved methodology in WCAP-14040-NP-A for P/T curves ensures the requirements of 10 CFR 50 Appendix G are met. Unit 1 and Unit 2 plant specific P/T curve methodology (WCAP-15878 and WCAP-15047) is based on WCAP-14040-NP-A. Therefore, 10 CFR 50 Appendix G continues to be met.

Based on the considerations above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will continue to be conducted in accordance with the site licensing basis, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

In conclusion, CNP has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any regulatory requirements or criteria.

4.2 No Significant Hazards Consideration

This application for amendment to the CNP TS proposes to revise TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits", to address an issue regarding the applicability of Figures 3.4.3-1 "Reactor Coolant System Pressure versus Temperature Limits – Heatup Limit, Criticality Limit, and Leak Test Limit (Applicable for service period up to 32 EFPY)" and 3.4.3-2 "Reactor Coolant System Pressure versus Temperature Limits – Various Cooldown Rates Limits (Applicable for service period up to 32 EFPY)" during vacuum fill of the RCS.

TS Figures 3.4.3-1 and 3.4.3-2 provide the RCS pressure versus temperature limits for various modes of reactor operation. These curves specify safe zones of reactor operation under varying RCS pressure and temperature conditions.

As required by 10 CFR 50.91(a), the CNP analysis of the issue of no significant hazards consideration is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. There are no physical changes to the plant being introduced by the proposed changes to the heatup and cooldown limitation curves. The proposed changes do not modify the RCS pressure boundary. That is, there are no changes in operating pressure, materials, or seismic loading. The proposed changes do not adversely affect the integrity of the RCS pressure boundary such that its function in the control of radiological consequences is affected.

Therefore, it is concluded that the proposed amendment does not involve a significant increase in the probability or the consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Further, the proposed changes to the heatup and cooldown limitation curves do not affect any activities or equipment other than the RCS pressure boundary and do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Consequently, the proposed changes do not create the possibility of a new or different kind of accident, from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed TS changes do not involve a significant reduction in the margin of safety. The revised heatup and cooldown limitation curves and low-temperature overpressure protection limits are established in accordance with current regulations and the ASME B&PV Code 1995 edition with 1996 Addenda. These proposed changes are acceptable because the ASME B&PV Code maintains the margin of safety required by 10 CFR 50.55(a). Because operation will be within these limits, the RCS materials will continue to behave in a non-brittle manner consistent with the original design bases.

Therefore, the proposed amendment does not involve a significant reduction in margin of safety.

4.3 Precedence

On March 5, 2014, the NRC issued an amendment (Reference 7) to the Indian Point Nuclear Generating Unit 2 to revise the P/T Limit curves, which included a note to allow operation of the RCS in a vacuum condition. CNP Unit 1 and Unit 2 have similar RCS operating P/T requirements in that they are both Westinghouse design pressurized water reactors. Additionally, the methodology used to develop the CNP and Indian Point P/T curves is based on the NRC approved WCAP-14040-NP-A.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the NRC's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. I&M concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.0 ENVIRONMENTAL CONSIDERATION

I&M has evaluated this LAR against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. I&M has determined that this LAR meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR Part 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 4.2, the proposed TS change does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

This LAR will not change the types or amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

This LAR will not increase the individual or cumulative occupational radiation exposure.

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. WCAP-15878, "D.C. Cook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation for 40 Years and 60 Years, Revision 0 (December 2002)."

2. WCAP-15047, "D. C. Cook Unit 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation, Revision 2 (May 2002)."

3. WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, Revision 2 (January 1996)."

4. Westinghouse Letter MCOE-LTR-14-17 "Applicability of the Pressure-Temperature Limit Curve Figures During Vacuum Refill of the RCS in Mode 5 for Westinghouse and CE NSSS Plants", Revision 0, dated March 3, 2014 (Westinghouse Proprietary Class 2).

5. 10CFR50 Appendix G – Fracture Toughness Requirements.

6. Letter from D. V. Pickett, NRC, to Vice President, Operations, Entergy Nuclear Operations, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: Pressure-Temperature Limit Curves and Low Temperature Over Pressure Requirements (TAC No. MF0634)," dated March 5, 2014 (ADAMS Accession No. ML14045A248).

7. Westinghouse SECL-96-226, Reactor Coolant Vacuum Refill System Final SECL, Revision 0, dated January 10, 1997.

Enclosure 3 to AEP-NRC-2014-24

Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages
Marked To Show Proposed Changes

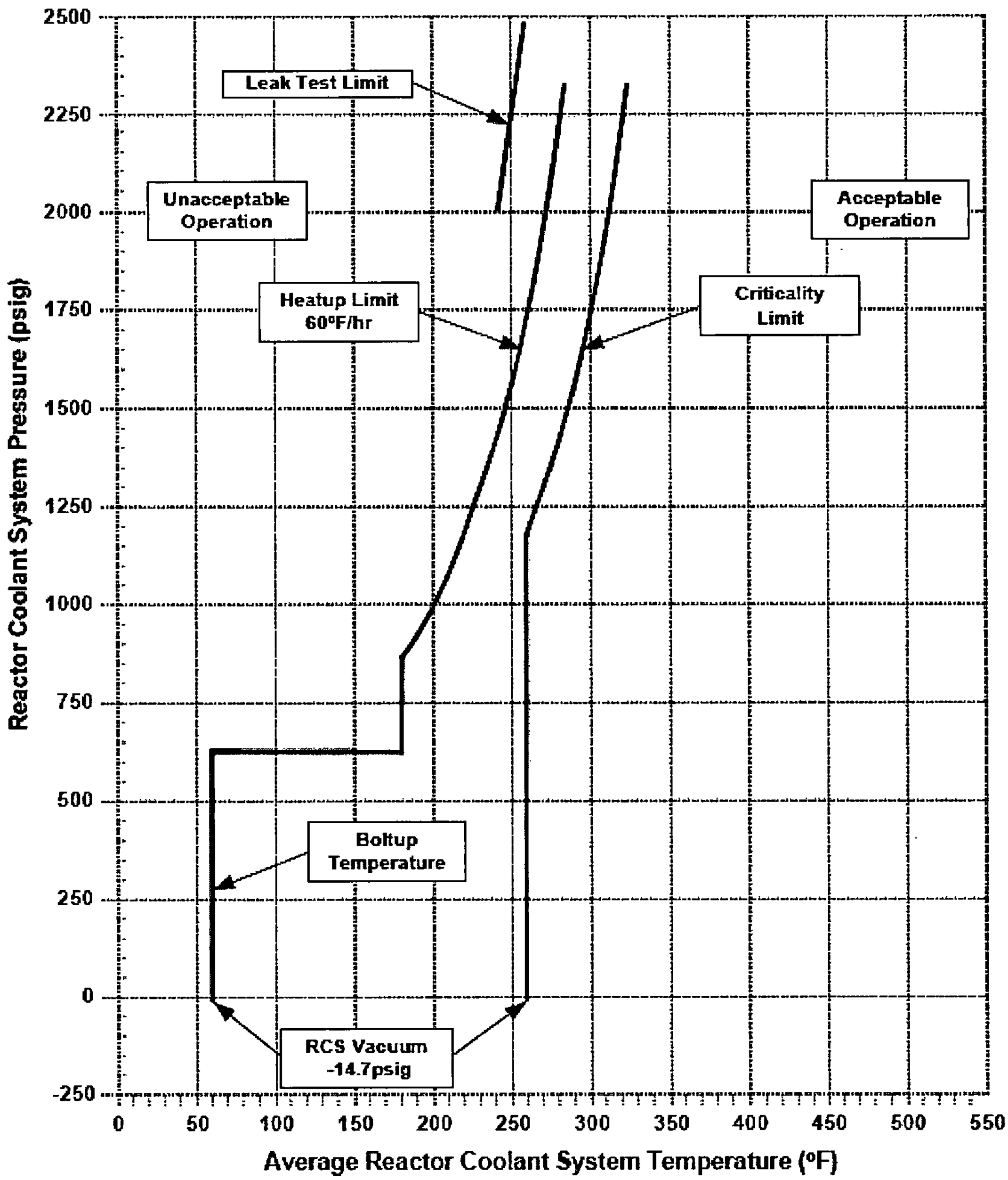

Figure 3.4.3-1 (page 1 of 1)
Reactor Coolant System Pressure versus Temperature Limits -
Heatup Limit, Criticality Limit, and Leak Test Limit
(Applicable for service period up to 32 EFPY)

and during vacuum fill

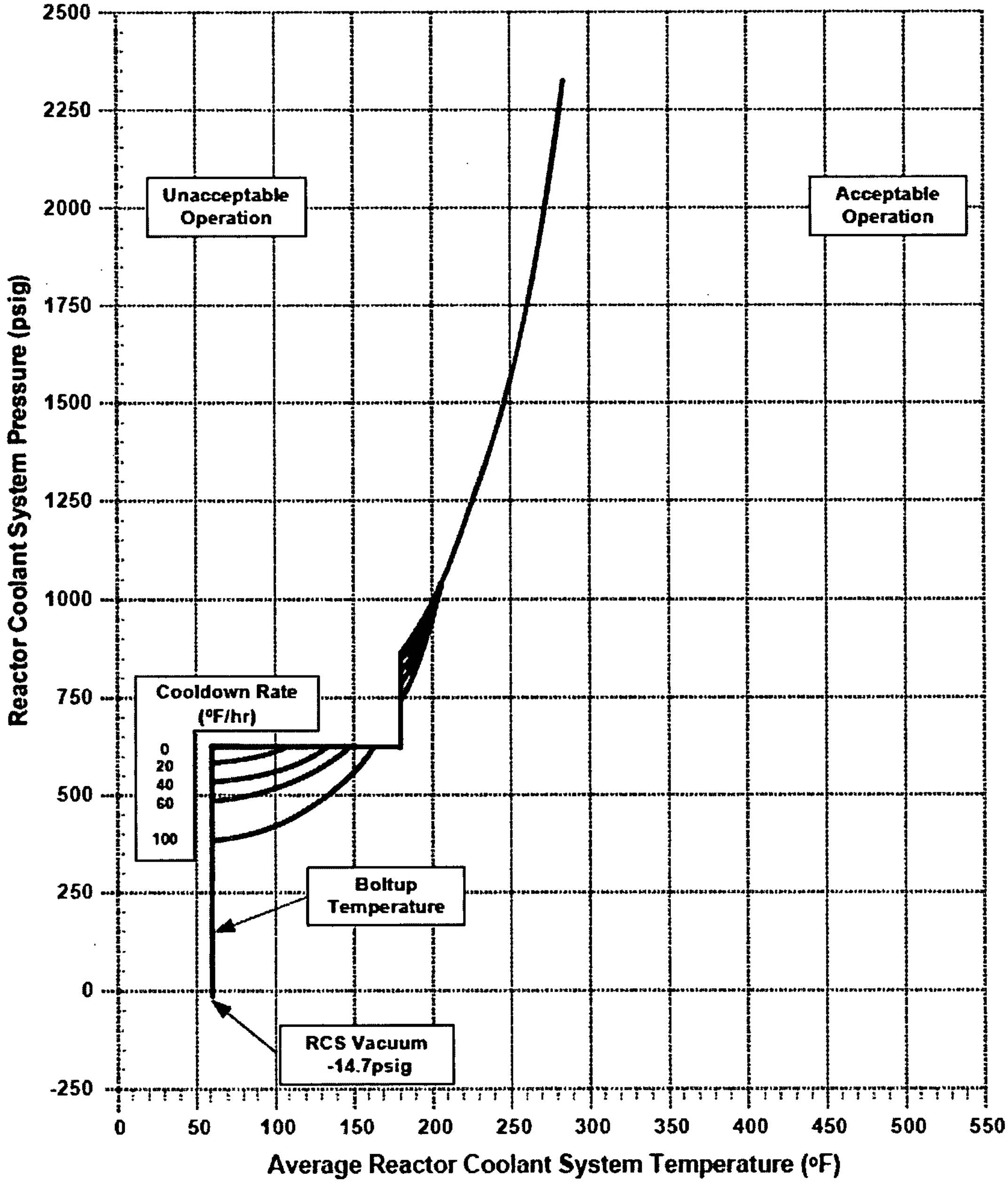

Figure 3.4.3-2 (page 1 of 1)
Reactor Coolant System Pressure versus Temperature Limits -
Various Cooldown Rates Limits
(Applicable for service period up to 32 EFPY)

and during vacuum fill

Enclosure 4 to AEP-NRC-2014-24

Donald C. Cook Nuclear Plant Unit 2 Technical Specification Pages
Marked To Show Proposed Changes

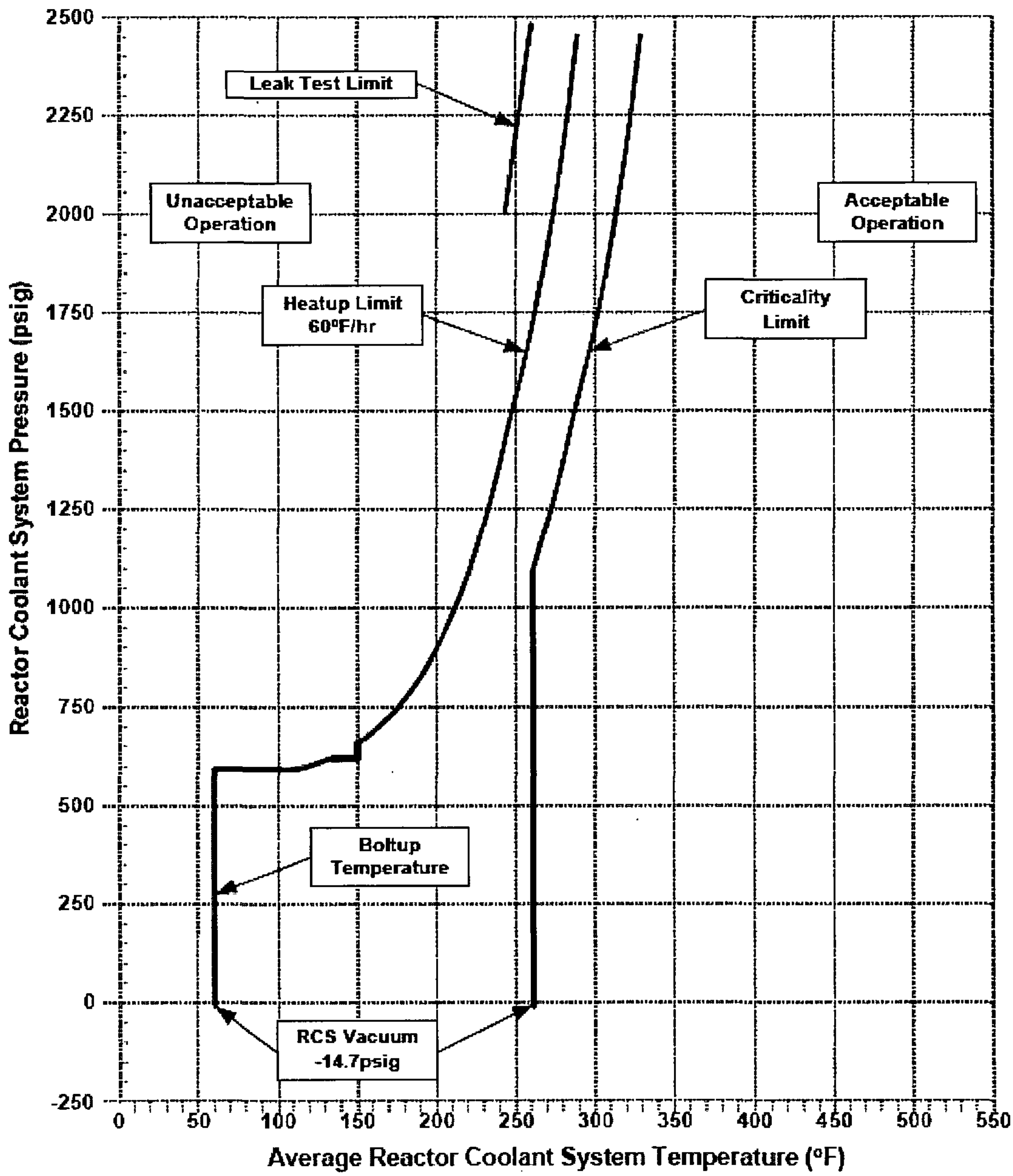

Figure 3.4.3-1 (page 1 of 1)
Reactor Coolant System Pressure versus Temperature Limits -
Heatup Limit, Criticality Limit, and Leak Test Limit
(Applicable for service period up to 32 EFPY)

and during vacuum fill

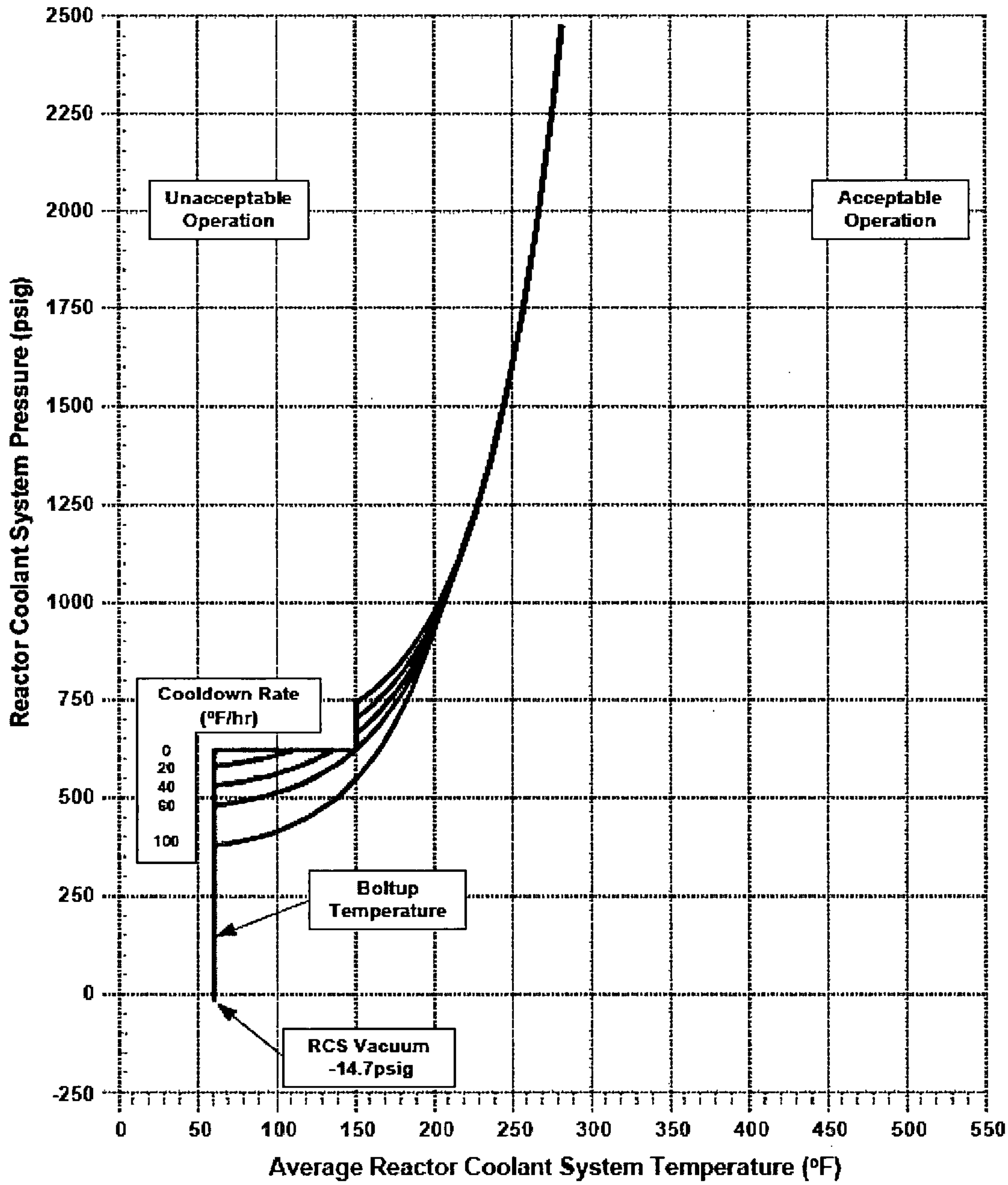

Figure 3.4.3-2 (page 1 of 1)
Reactor Coolant System Pressure versus Temperature Limits -
Various Cooldown Rates Limits
(Applicable for service period up to 32 EFPY)

and during vacuum fill

Enclosure 5 to AEP-NRC-2014-24

Donald C. Cook Nuclear Plant Unit 1 Technical Specification Bases Pages Marked to Show Proposed Changes for Information

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

This LCO contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, criticality, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region. Vacuum fill of the RCS is performed in Mode 5 under sub-atmospheric pressure and isothermal RCS conditions. Vacuum fill is an acceptable condition since the resulting pressure/ temperature combination is reflected on the operating limits provided in Figures 3.4.3-1 and 3.4.3-2.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, criticality, and ISLH testing; and

b. Limits on the rate of change of temperature.

Enclosure 6 to AEP-NRC-2014-24

Donald C. Cook Nuclear Plant Unit 2 Technical Specification Bases Pages Marked to Show Proposed Changes for Information

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

This LCO contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, criticality, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region. Vacuum fill of the RCS is performed in Mode 5 under sub-atmospheric pressure and isothermal RCS conditions. Vacuum fill is an acceptable condition since the resulting pressure/ temperature combination is reflected on the operating limits provided in Figures 3.4.3-1 and 3.4.3-2.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, criticality, and ISLH testing; and

b. Limits on the rate of change of temperature